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BERMAN, H.S.		
CARNIVAL, G.J.		
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CORDOVA, R.C.		
DAVIS, J.G.	X	X
FERRERA, D.W.		
FRANZ, W.A.		
HANNI, B.J.		
HEALY, T.J.		
HEDAHL, T.G.	X	X
HILBIG, J.G.		
HUTCHINS, N.M.		
KELL, R.E.	X	X
KIRBY, W.A.		
KUESTER, A.W.		
MAHAFFEY, J.W.		
MANN, H.P.	X	
MARX, G.E.		
McKENNA, F.G.		
MORGAN, R.V.		
PIZZUTO, V.M.		
POTTER, G.L.		
SANDLIN, N.B.		
SATTERWHITE, D.G.		
SCHUBERT, A.L.		
SETLOCK, G.H.		
STIGER, S.G.	X	X
SULLIVAN, M.T.	X	X
SWANSON, E.R.	X	X
WILKINSON, R.B.	X	
WILSON, J.M.	X	
Bennett W	X	X

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Rocky Flats Office

EG&G ROCKY FLATS PLANT CORRES. CONTROL

MAR 3 0 1994
SPA:DWS:03386

Westinghouse Electric Corporation Government Operations Radiological Risk Acceptance Guidelines For High Level Waste Tanks

John G. Davis, Assoc. General Manager, Standards, Audits & Assurance, EG&G Rocky Flats, Ir
Wayne M. Wright, Manager, Quality Assurance, Wackenhut Services, Inc., RF

Attached is a copy of the Westinghouse Electric Corporation Government Operations Radiologi Rick Acceptance Guidelines For High Level Waste Tanks. This manual is provided to share wh other sites are doing to improve their operations. This is an opportunity for us to share their expertise, talent, and technology to improve the operations at the Rocky Flats Plant.

Please review this manual and provide any actions that your organization takes to the undersigned. If you have information that should be shared with other sites, please provide that information to the undersigned.

000027327

Dero W. Sargent
Dero W. Sargent, Director
Standards, Performance, & Assurance

Attachment

cc w/o Att:
M. Silverman, OOM, RFO
L. Smith, OOM, RFO
H. Mann, EG&G, RF
W. Gillison, WSI, RF

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Radiological Risk Acceptance Guidelines For High-Level Waste Tanks

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RADIOLOGICAL RISK ACCEPTANCE GUIDELINES FOR HIGH-LEVEL WASTE TANKS

I. SUMMARY

This document starts with the DOE Nuclear Safety Policy (SEN-35-91) and the premise stated in that policy that the general public should bear no significant additional risk from operation of DOE nuclear facilities above the risks to which members of the general population are normally exposed. Public radiological risk guidelines are then developed which are consistent with this premise of the Nuclear Safety Policy. The public radiological risk guidelines are based on a maximum exposed individual concept rather than a population risk, and provide a method for an accident-by-accident comparison of the individual accident risks. It is then demonstrated that the radiological risk guidelines provide a sufficient, but not necessary, condition for compliance with the quantitative safety goals of the Nuclear Safety Policy.

An analogous philosophy to the Nuclear Safety Policy is then applied to collocated workers at a DOE site to develop collocated worker radiological risk guidelines. Members of the on-site public and facility workers are also considered. The application of the radiological risk guidelines in the on-going iterative safety analysis - design process is discussed in relation to protection of the public and workers.

It should be noted that the radiological risk guidelines are to be used for individual accident risk comparisons and are not related to the total risk of a facility. However, as already noted, the document demonstrates that compliance with the radiological risk guidelines is a sufficient condition to demonstrate that the total risk complies with the quantitative goals of the Nuclear Safety Policy.

II. INTRODUCTION

DOE Orders require that DOE contractors responsible for design, construction and operation of DOE non-reactor nuclear facilities, including high-level waste tank facilities, conduct safety analyses. As specified in these DOE Orders, the safety analysis reports required for DOE nuclear facilities must contain the analyses needed to authorize facility operations. A major element of the safety analysis process consists of determining and comparing the estimated risk from postulated radiological releases to specific risk acceptance criteria or guidelines.

The DOE issued their Nuclear Safety Policy for implementation on September 9, 1991 (SEN-35-91). A major premise of the Nuclear Safety Policy is that the general public should be protected such that no individual would bear significant additional risk to health and safety from operation of DOE owned nuclear facilities above the risks to which members of the general population are normally exposed. In order to achieve this general premise, the Nuclear Safety Policy provides two quantitative safety goals, which are discussed in Section VI. The DOE safety goal for prompt fatalities is applied to an average individual located within one mile of the site boundary. The safety goal for latent cancer effects is applied to a population within ten miles from the site boundary. In both cases, the goals are for an average individual who is part of the general public.

Total integrated risk may be compared directly to the DOE Nuclear Safety Policy if a probabilistic risk assessment has been conducted. Where a probabilistic risk assessment does not exist, a finite set of accidents is generally analyzed as a basis for risk acceptance. The radiological risk acceptance guidelines developed in this document apply to individual accidents for which discrete consequences and frequencies have been estimated. The set of individual accidents analyzed and compared with these guidelines should contain the risk dominant sequences.

This document presents risk acceptance guidelines, which if satisfied, support the conclusion that there is no undue risk to the public and workers from potential releases of radioactive material. The risk acceptance guidelines were derived from current DOE Orders, Directives and Policies, EPA and NRC Federal Regulatory Requirements, commercial nuclear industry practices, and applicable national and international radiation protection recommendations. Specific information considered and utilized in the development of the risk acceptance guidelines include publications of the:

International Commission on Radiological Protection (ICRP), specifically publications 26, 30, 37, 45, and 60.

National Council on Radiation Protection and Measurements (NCRP), specifically report number 91.

National Research Council, specifically the BEIR IV and V reports.

Environmental Protection Agency (EPA), specifically 40CFR61.

Department of Energy (DOE) Orders and Policies, specifically DOE Orders 5400.5, 5480.11, 5481.1B, 6430.1A, 5500.3A, 5480.23, and SEN-35-91.

Code of Federal Regulations, specifically 10CFR100, 10CFR72, and other appropriate Nuclear Regulatory Commission NUREG documents.

III. DEFINITIONS

Specific definitions were taken from DOE Orders, specifically Attachment 2 to DOE Order 5500.1B, which provides definitions for all DOE Orders in the 5500 series (Emergency Preparedness), and DOE Order 5480.11. From the definitions taken from DOE Orders other definitions were developed in order to make the context of the radiological risk acceptance guidelines more understandable.

A. DEFINITIONS TAKEN FROM DOE ORDERS

<u>Site</u> The area over which DOE has access control authority. This includes any area that has been designated as a National Security Area.

<u>Off-site</u> The area beyond the boundaries of the site.

<u>On-site</u> The facility or site area over which the Lead Federal Agency has access control authority. The on-site area includes any area that has been established as a National Defense Area or National Security area.

Facility — Any equipment, structure, system, process, or activity that fulfills a specific purpose.

B. DEFINITIONS DEVELOPED FROM DOE ORDERS

Site Boundary — The perimeter of the area over which DOE has access control authority.

Off-site Public — Those members of the general public who exist or reside outside of the area over which DOE has access control authority (site).

On-site Public — Those members of the general public who may for whatever reason be temporarily on the DOE site (i.e. on the site for a reasonably short period of time). This includes those members of the general public who may be visitors to the site, at a visitors center, or traversing the site on a public road or water way.

Facility Boundary — The perimeter of the area which a facility occupies. Principal considerations in defining a facility or activity boundary should be a fence or other physical barrier provided for process, security or emergency preparedness purposes. The boundary should encompass all buildings, structures, support equipment, and auxiliary systems that support a common mission. A complex may be considered as a single facility if all buildings are physically adjacent, under common management, contribute to a common programmatic mission, and are included in a common emergency response plan.

Facility Workers — Those individuals who work in a specific facility, who receive detailed job related training relative to the operations of the facility, whose job requirements identify the risks associated with working in the facility, and who are protected by special emphasis contained in ALARA programs.

Collocated Workers — Those government and/or contractor employees who are on the DOE site but who are not workers in the facility being considered. (They may be workers in a facility which is not being analyzed.) This includes individuals who have access to the site by virtue of a concession contract, service contract, delivery contract, consultant contract, or construction contract. This also includes those individuals visiting the site who have been provided training specified in DOE Order N5480.6 (Chapter 6). It is understood that these individuals have accepted a larger risk than the general public by being on the site, but who may not have accepted the risk associated with the operation of a specific facility.

ALARA (DOE 5480.11 and N5480.6)

An approach to radiological control to manage and control exposures (individual and collective) to the work force and to the general public at levels as low as is reasonable, taking into account social, technical, economic, practical and public policy considerations. ALARA is not a dose limit but a process that has the objective of attaining doses as far below the applicable controlling limits as is reasonably achievable.

ALARA is the process of optimizing the radiological risk of a facility or an activity which already meets regulatory radiation dose limits. This evaluation which includes consideration of social, technical, economic, practical and public policy will optimize the effectiveness of the design to limit potential radiation dose to the lowest reasonably achievable level.

Maximum Exposed Off-Site Individual

A hypothetical individual located at the site boundary in the direction of the worst-case atmospheric dispersion factor (χ/Q) who remains at the plume center-line for the duration of a postulated radiological release.

Maximum Exposed Collocated Worker

A hypothetical individual located in a nearby on-site facility in the direction of the worst-case atmospheric dispersion factor (χ/Q) who remains at the plume center-line.

IV. RISK ACCEPTANCE GUIDELINES

The off-site and collocated worker risk acceptance guidelines are presented in this section followed by a detailed discussion of their technical bases and relation to the DOE Nuclear Safety Policy.

A. OFF-SITE PUBLIC RISK ACCEPTANCE GUIDELINE

The off-site public risk acceptance guideline is presented in Figure 1 where the logarithm of accident consequence is graphed as a function of the logarithm of the release accident frequency. The frequency range is divided into two regions: a lower region (1×10^{-6}/yr to 1×10^{-2}/yr) of accidents not expected to occur during facility lifetime (unlikely), and a higher region (1×10^{-2}/yr to 1/yr) of accidents which may be expected to occur during a facility lifetime (anticipated). This division uses features from Brynda et al., 1981, 1986, and Elder et al., 1986.

The lowest frequency considered is 1×10^{-6}/yr. Accidents with lower frequencies are not considered credible (Elder et al., 1986). At this frequency the maximum dose equivalent to an off-site individual is 25 rem (0.25 Sv). At the crossover between unlikely and anticipated accidents (frequency = 1×10^{-2}/yr), the maximum dose equivalent allowed to an off-site individual is

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0.5 rem (0.005 Sv) or 500 mrem. A straight line connects the 25 rem and 0.5 rem points. At the dividing point between normal operations and accidents (frequency = 1/yr), the maximum dose equivalent allowed to an off-site individual is 0.01 rem (0.1 mSv) or 10 mrem. A straight line connects the 500 mrem and 10 mrem points.

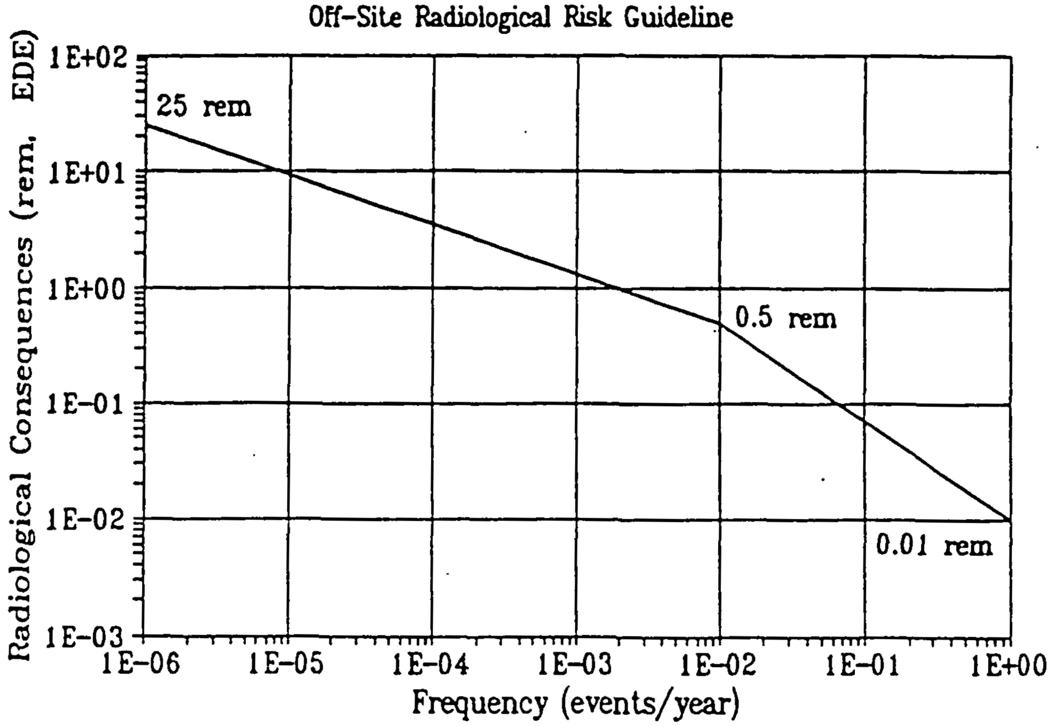

Figure 1. Off-Site Guideline

Note: This guideline is to be used to compare the consequences and frequency of single accidents.

B. ON-SITE Collocated WORKER RISK ACCEPTANCE GUIDELINE

The on-site collocated worker risk acceptance guideline is shown in Figure 2. At a frequency of 1×10^{-6}/yr the maximum dose equivalent allowed to an on-site individual is 100 rem (1 Sv). At the dividing point between unlikely and anticipated accidents (frequency = 1×10^{-2}/yr), the maximum dose equivalent allowed to an on-site individual should not exceed 5 rem (0.05 Sv). A straight line connects the 100 rem and 5 rem points. At the frequency of 1×10^{-1}/yr, the maximum dose to an on-site individual should not exceed 1 rem (0.01 Sv). A straight line connects the 5 rem and 1 rem points. For frequencies from 1×10^{-1}/yr to 1/yr, normal radiation practices are assumed to be in place to control radiation doses to within DOE-prescribed limits.

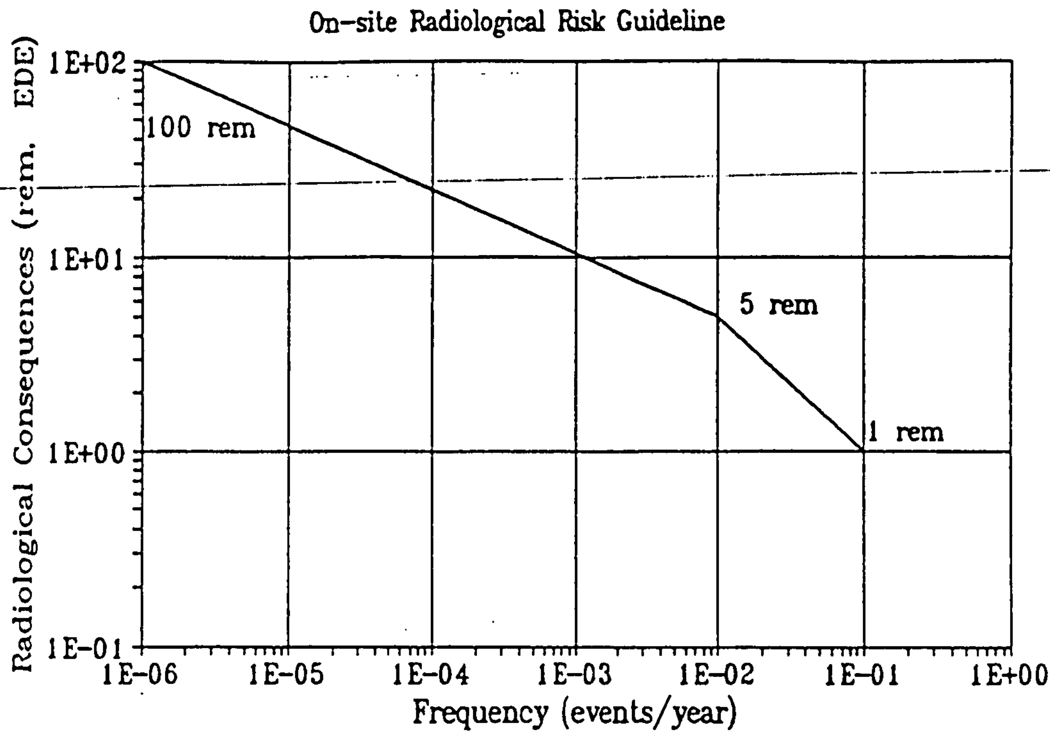

Figure 2. On-Site Guideline For Co-located Workers
Note: This guideline is to be used to compare the consequences and frequency of single accidents.

V. TECHNICAL BASES FOR RISK ACCEPTANCE GUIDELINES

A. GENERAL

ICRP-60 states that the simplest way of dealing with potential exposures to ionizing radiation is to consider the overall individual probability of attributable death from cancer. A risk value is defined as the product of an expected dose, the frequency of incurring that dose and the lifetime conditional probability of attributable death from that dose, if it were received. This definition will be used in this discussion.

The public accident frequency is taken from *Accident Facts (National Safety Council, 1990)*. The information cites fatalities for all accidents to be 3.9 x 10^{-4} prompt fatalities/(person-yr). From this, the DOE nucle10
safety goal of 0.1% of prompt fatalities converts to a risk of 4 x 10^{-7} prompt fatalities/(person-yr) (rounded). The cancer fatality rate in the United States is 1.93 x 10^{-3} fatalities/(person-yr). Thus, the DOE nuclear safety goal for latent cancer fatalities converts to a risk of 2 x 10^{-6} latent cancer fatalities/(person-yr) (rounded).

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Recent estimates of cancer hazard from ionizing radiation are presented in ICRP-60, BEIR V, and NCRP-91. The probability of fatal cancer induction to the public of all ages is estimated to be 5 x 10^{-4} cancer fatalities/rem (low dose, low dose-rates). The probability of fatal cancer induction from occupational exposure is estimated to be 4 x 10^{-4} cancer fatalities/rem. These values are taken from ICRP-60 which presents the most recent analysis of radiation risk data. A compilation of known public and occupational dose limit recommendations and regulations is provided in the Appendix.

In the discussion that follows, in order to emphasize the risk for specified consequence-frequency pairs, the concept of an individual accident risk to the maximum exposed person will be specified as R_{iME}. For example, an effective dose to the maximum exposed individual of 0.5 rem at a frequency of 1 x 10^{-2}/yr will have a R_{iME} = 5 x 10^{-3} rem/yr. This individual accident risk is provided for illustrative purposes only and should not be taken to be the total risk.

B. OFF-SITE PUBLIC

For accident frequencies that may occur once per year, the risk acceptance guideline was taken to be 10 mrem at this frequency (R_{iME} = 1 x 10^{-2} rem/yr). This guideline was derived by considering that any radiation released in a year from the operation of a DOE facility should give an effective dose equal to or less than the effective dose allowed to the public. First, DOE Order 5400.5 places an airborne emissions limit, for the airborne pathways only, of 10 mrem/yr which "applies off-site where members of the public reside and abide." The total dose allowed by DOE Order 5400.5 is 100 mrem/year. Second, the EPA limit on airborne releases allowed to any member of the public is 10 mrem/year (40 CFR 61.92). Third, the ICRP recommended a public annual limit of 100 mrem/year. A reduction of a factor of ten from the ICRP limit seemed appropriate, which would conform to the DOE 5400.5 and EPA limits.

The accident frequency which divides the two sections of the suggested curve is 1 x 10^{-2}/yr, which is the approximate dividing point between accidents which may occur more or less than once in an individual or facility lifetime. This frequency was used in previously suggested guides to divide anticipated accidents from unlikely accidents (Brynda et al., 1981, 1986), (Elder et al., 1986). The consequence level where the two sections intersect was given special attention.

For those accidents which may release radiation once in the lifetime of a facility (taken to be 1 x 10^{-2}/yr), the effective dose should be equal to or less than the effective dose allowed to the public for infrequent exposure. Thus, the risk acceptance guideline of an effective dose of 0.5 rem at a frequency of 1 x 10^{-2}/yr (R_{iME} = 5 x 10^{-3} rem/yr) was selected on the following basis. First, DOE Order 5400.5 allows an effective dose equivalent of up to 0.5 rem/yr to the public in special cases if the average lifetime dose is less than 0.1 rem/yr. Second, ICRP 60 and 45, and NCRP 91, recommend a subsidiary effective dose limit of 0.5 rem/yr if the dose averaged over a lifetime does not exceed 0.1 rem per year. Thus, it is considered conservative to apply the 0.5 rem value at an accident frequency of 1 x 10^{-2}/yr for the purpose of risk acceptance guidelines. (This gives a risk of 5 mrem/yr.) Revision 1 of Brynda, et al., also implied this same limit of 0.5 rem at an accident frequency of 1 x 10^{-2}/yr.

The maximum radiological off-site dose of 25 rem at the frequency of 1×10^{-6}/yr ($R_{iME} = 2.5 \times 10^{-5}$ rem/yr) is specified in the 10 CFR 100 siting criteria for nuclear power plants (10 CFR 100.11). The siting criteria emphasized that this number was to be used for accidents "of exceedingly low probability of occurrence and low risk of public exposure to radiation." DOE Order 6430.1A restated the 25-rem effective dose as a limit for siting DOE nonreactor nuclear facilities. Both Brynda, et al. and Elder, et al. used the same limit of 25 rem at an accident frequency of 1×10^{-6}/yr. Thus, the off-site risk acceptance guideline used this effective dose (25 rem) at the lowest frequency of credible accidents ($R_{iME} = 2.5 \times 10^{-5}$ rem/yr).

Further consideration was provided regarding the Off-site Public Risk Acceptance Guidelines as follows. The NRC noted in their proposed policy statement titled "Below Regulatory Concern: Policy Statement" (U. S. NRC, 1990), that incremental annual risks in the range of 5×10^{-6} to 5×10^{-7} fatalities/yr are well within the range of risk commonly accepted by the public. The NCRP (NCRP-91) supported this range of risk as negligible public risk from industrial operations. These population risks correspond to radiation doses in the range from 1 mrem/yr to 10 mrem/yr, using the ICRP-60 average lifetime risk factor of 5×10^{-4} cancer fatalities/rem. The risk acceptance guideline for a maximum-exposed individual falls within or below this range of average acceptable public risk. Thus, the guideline is conservative with respect to the risks normally accepted by the public over the entire range of credible accident frequencies.

C. Collocated WORKERS

Normal radiation protection practices are assumed to be in place at all DOE facilities to control radiation releases from normal operations to within prescribed limits. This is assumed to include accident situations that may occur with frequency from once per year (1/yr) to once in 10 years (0.1/yr). DOE Order N5480.6 (Radiological Control Manual) establishes a goal of 0.5 rem/yr for occupational exposure from normal operations for most facilities.

For accident frequencies that may occur on the order of once in 10 years, the risk acceptance guideline for collocated workers was taken to be 1 rem ($R_{iME} = 1 \times 10^{-1}$ rem/yr). This guideline was derived by considering that any radiation released from the operation of a DOE facility on the order of once in 10 years should give an effective dose equal to or less than the effective dose allowed for workers. The ICRP-60 recommends an occupational exposure limit of an effective dose of 2 rem/yr averaged over 5 years (10 rem in 5 years) providing the effective dose does not exceed 5 rem in a single year. DOE Order 5480.11 sets a 5 rem/yr limit for occupational exposures. The design basis specified by DOE Order 6430.1A is 1 rem/yr. Thus, in order to provide extra protection to the collocated workers, an exposure of 1 rem for frequencies of the order of once per ten years was selected as the risk acceptance guideline.

For those accidents which may result in the release of radiation once in the lifetime of a facility (again taken to be 1×10^{-2}/yr), the effective dose should be equal to or less than the effective dose allowed to workers for infrequent exposure. Thus, the risk acceptance guideline of an effective dose of 5 rem at a frequency of 1×10^{-2}/yr ($R_{iME} = 5 \times 10^{-2}$ rem/yr) was selected on the following basis. DOE Order 5480.11 allows a limiting value of 5 rem effective dose received in any year by an occupational worker. As noted earlier, the ICRP-60 recommends an occupational exposure limit of an effective dose of 2 rem/yr averaged over 5 years providing the effective dose does not exceed 5 rem in a single year. Thus, an exposure of 5 rem for the collocated worker is not unreasonable for accidents which occur less than once in a lifetime.

The maximum consequence from the risk acceptance guideline is an effective dose of 100 rem at a frequency of 1×10^{-6}/yr ($R_{iME} = 1 \times 10^{-4}$ rem/yr). This effective dose limit was derived as follows. The probability of fatal cancers from high dose and high dose rates is about 1×10^{-3} fatalities/rem, (ICRP-60, p. 133). Combining this factor with the annual fatality for safe activities, 1×10^{-4} fatalities/yr, with the assumption that the risk should be 0.1% of this value, produces an effective radiological dose of 100 rem at a frequency of 1×10^{-6}/yr.

Further consideration was provided regarding the On-site Collocated Worker Risk Acceptance Guidelines as follows. The all-industry occupational fatality risk is approximately 1×10^{-4} fatalities/yr. The occupational risk for industries considered "safe" is one or fewer fatality per 10,000 workers per year (NCRP-91, June 1987, p. 8). The probability of fatal cancer induction after low dose, low dose-rate irradiation of a worker population is 4×10^{-4} fatalities/rem, (ICRP-60, November 1990, p. 133). With this factor, the risks of 1 rem at 1×10^{-1}/yr, 5 rem at 1×10^{-2}/yr and 100 rem at 1×10^{-6}/yr equate to 4×10^{-5} fatalities/yr, 2×10^{-5} fatalities/yr, and 4×10^{-8} fatalities/yr, respectively. Thus, the on-site risk acceptance guideline is lower than the risk for safe industries, even though the guideline is intended to apply to a maximum-exposed rather than an average collocated worker. Therefore, the on-site risk acceptance guidelines are conservative.

D. FACILITY WORKERS

DOE occupational radiation limits identified in DOE Order 5480.11, "Radiation Protection For Occupational Workers," apply to all workers directly engaged in the facility operations. It is DOE's policy that facilities are operated such that radiation exposures from both internal and external sources are maintained within an effective dose of 5 rem/yr and as far below this limit as reasonably achievable. DOE has established a maximum administrative control level of 2 rem/yr per person (DOE Order N5480.6, Radiological Control Manual) and a goal of 0.5 rem/yr for occupational exposure from normal operations. Thus, workers are protected by application of the As Low As Reasonably Achievable (ALARA) philosophy. Hence, no radiological risk acceptance guideline was developed for facility workers.

E. ON-SITE PUBLIC

The following position is provided relative to the on-site public. Those members of the general public who may be on the site for a reasonably short period of time should be afforded the same measure of protection as the off-site public. Therefore, the off-site public risk acceptance guidelines also apply to the on-site public. However, consideration needs to be given in terms of the parameters used to calculate the radiation exposure which may be received by the on-site public. The radiation exposure to an individual is proportional to the dispersion factor (χ/Q, which decreases logarithmically with distance) and the exposure time. Therefore, in order to comply with the off-site public risk acceptance guidelines for the on-site public, it is necessary that the on-site public be exposed to radiation for the shortest possible time period. Thus, provisions should be made that public visiting the site are escorted and can be quickly moved from the path of released radiation, or if not escorted, the on-site public can be quickly evacuated from the site.

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VI. RELATION OF GUIDELINES TO DOE NUCLEAR SAFETY POLICY

A. DOE QUANTITATIVE SAFETY GOALS

The premise of the U.S. Department of Energy (DOE) Nuclear Safety Policy is that the general public should bear no significant additional risk from operation of DOE nuclear facilities above the risks to which members of the general population are normally exposed. The policy provides general statements in the areas of management involvement and accountability, providing technically competent personnel, oversight and self-assessment, and promoting a safety culture.

The Nuclear Safety Policy also provides two quantitative safety goals to limit the risks of fatalities associated with its nuclear operations. These goals are the same as those established for commercial nuclear power plants by the NRC and should be viewed as "aiming points for performance." These goals are:

- The risk to an average individual in the vicinity of a DOE nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatalities resulting from other accidents to which members of the population are generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary.

- The risk to the population in the area of a DOE nuclear facility for cancer fatalities that might result from operations should not exceed one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes. For evaluation purposes, individuals are assumed to be located within ten miles of the site boundary. These quantitative DOE safety goals may be expressed in terms of the number of expected prompt fatalities and latent cancer fatalities by using information from the Accident Facts (National Safety Council 1990). The total U. S. death rate from all accidents is 39 prompt fatalities per 100,000 people every year. This represents approximately 4×10^{-4} prompt fatalities per person per year. Multiplying this value by the 0.1% as noted in the first goal yields

$$4 \times 10^{-7} \text{ prompt fatalities/person year.}$$

The total death rate from all cancers is 196 latent cancer fatalities per 100,000 people every year, or approximately 2×10^{-3} latent cancer fatalities per person per year. Multiplying this value by the 0.1% as noted in the second goal yields

$$2 \times 10^{-6} \text{ latent cancer fatalities/person year.}$$

With this information, the DOE safety goals may be expressed as follows.

- The risk of a prompt fatality for an average individual assumed to be within 1 mile of the site boundary should be less than 4×10^{-7} prompt fatalities per year (or 1 chance of a prompt fatality in 2.5 million years).

- The risk of a latent cancer fatality for the general public located within 10 miles of the site boundary should be less than 2×10^{-6} latent cancer fatalities per average individual per year (or 1 chance of latent cancer fatality in a half million years).

Both the NRC and DOE safety goals for prompt fatalities from an accident are intended to be applied to an average individual located near the facility of concern. Safety goals for latent cancer (long-term effects) are intended to be applied to populations exposed to all radioactive releases within a specific distance from the site boundary around the facility of concern. In both cases, the goal is for an individual who is part of the general public, and the hazard that causes the effect arises from the operation of a facility that could release radioactive materials. The risk posed by operation of a facility would be dominated by the potential release of radiation during off-normal accident conditions.

B. COMPLIANCE WITH DOE SAFETY GOALS

Compliance with the DOE safety goal can be demonstrated by completing a rigorous probabilistic risk assessment of an activity or facility to quantify both the frequency of occurrence and the consequence of each accident to determine the total risk. The total risk could then be compared with the quantitative statement of the safety goal, R_{SG}. For example, assume that there exists a complete set of N accident scenarios defined by S_j with frequency F_j and consequence C_j. The complete set of accidents is defined by

$$\{ S_j , F_j , C_j \} . \tag{1}$$

The total risk for this complete set of accident scenarios is given by:

$$R_T = \sum_{j=1}^{j=N} C_j F_j . \tag{2}$$

The calculated total risk, in units of either prompt fatalities or latent cancer fatalities per person per year could then be compared with the two quantitative safety goals. If

$$R_T \leq R_{SG} , \tag{3}$$

then the facility would comply with the safety goal.

For many DOE facilities and activities, a rigorous probabilistic risk assessment may not be warranted. In the Nuclear Safety Policy, DOE noted that the adoption of the safety goals should not be construed as a requirement to conduct probabilistic risk assessments. Therefore, this suggests that there may be other satisfactory criteria which may be used to assess whether or not a facility complies with the safety goals.

Consider the complete set of N accident scenarios defined by Equation (1), such that the total risk is given by Equation (2). Assume that each accident scenario is represented by a point $\{F_j,C_j\}$ in a two-dimensional consequence-frequency space. In most probabilistic risk assessments, it is common to divide the accident into groups based on comparable consequences. Because the risk acceptance guidelines were developed on the basis of consequence versus frequency, this demonstration will be based on dividing the frequency range into groups. Similar results are obtained if one divides the accident sequences into groups which have comparable consequences. Consider that the frequency domain in this two-dimensional space is divided into m frequency groups.

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The total risk can now be represented by:

$$R_T = \sum_{j=1}^{j=m} \sum_{k=1}^{k=L} F_{jk}\, C_{jk} \tag{4}$$

where the second summation is over the set of points in each frequency group, and the first summation is over the m frequency groups with L accident sequences in each group.

Within each frequency group, the frequency is nearly the same, or can be replaced by an average frequency assuming small frquency groups. Assume that for each set of consequences in each frequency group there exists a maximum consequence C_{jM} such that $C_{jk} < C_{jM}$ for each C_{jk} in the frequency group. Let L_j represent the number of consequences in the j-th frequency group. The total risk is then less than:

$$R_T < \sum_{j=1}^{j=m} F_j\, C_{jM}\, L_j \; . \tag{5}$$

We now define R_{SG} as the magnitude of the DOE quantitative safety goal for latent cancer fatalities (LCF), that is, $R_{SG} = 2 \times 10^{-6}$ LCF/person-year.

Then, $R_T < R_{SG}$ if

$$\sum_{j=1}^{j=m} F_j\, C_{jM}\, L_j = R_{SG} \; . \tag{6}$$

Thus, the criteria required for the total risk, R_T, to be less than or equal to the magnitude of the safety goal, R_{SG}, are:

$$R_T < R_{SG} \quad \text{if}$$

(1) the calculated consequence for each accident is less than the maximum consequence for each group frequency, (i.e. $C_{jk} \leq C_{jM}$ for each accident k in frequency group j),

(2) $\sum_{j=1}^{j=m} F_j\, C_{jM}\, L_j = R_{SG}$,

where $R_{SG} = 2 \times 10^{-6}$ LCF/person-year, and

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(3) $$\sum_{j=1}^{j=m} L_j = N$$, the sum of the accident sequences in each frequency group, L_j,

summed over all frequency group is equal to the total number of sequences, N.

The above criteria may be considered as representing a locus of points $\{F_j, C_{jM}\}$ in a two-dimensional frequency-consequence space where the consequence for each accident scenario in a specific frequency group is less than or equal to the maximum consequence for that frequency group. The criteria are sufficient conditions to show that the total risk is less than or equal to the magnitude of the safety goal, but are not necessary conditions. Such criteria would allow an accident-by-accident comparison of the calculated consequence with the maximum consequence for the specific frequency group. The objective will be to demonstrate that the proposed Radiological Risk Acceptance Guidelines satisfy the above criteria, and hence, can be used to demonstrate compliance with the DOE safety goals.

An example will help illustrate the criteria which has been identified. Consider an arbitrary set of risk dominant accident sequences where the frequencies and consequences have been determined. Such a set is illustrated in Figure 3 where, for example, 55 sequences (represented by the solid squares) have been graphed on a log-log graph in a two-dimensional frequency-consequence space. It is further assumed that the solid line in Figure 3 represents the radiological risk acceptance guideline, and that each accident sequence satisfies the guideline (i.e. lies below the guideline as illustrated in Figure 3). Now consider that the calculated consequence for each sequence is replaced by the appropriate maximum consequence for that frequency from the guideline. The set of resulting points are represented by the open squares in Figure 4 which lie on the guideline. Next consider that the frequency and consequence for each sequence are multiplied and the product is summed over all the sequences, in appropriate units. From Figure 4 it is obvious that the sum of the frequency-consequence products which represent the guideline (open squares) will be larger than the sum of the frequency-consequence products for the actual accident sequences (solid squares).

In this example of m=55 sequences (presented in Figures 3 and 4 appropriately converted from maximum exposed individual to average individual), the total risk due to the accident sequences is

$$\sum_{j=1}^{j=m} F_j\, C_j = 1.6 \times 10^{-8}\ LCF/person\text{-}year\ ,$$

while the sum of the frequency-consequence products which represent the radiological risk acceptance guideline is

$$\sum_{j=1}^{j=m} F_j\, C_{jM} = 1.2 \times 10^{-7}\ LCF/person\text{-}year\ .$$

This is the point of the criteria. If the calculated consequence-frequency pairs are less than the radiological risk acceptance guideline on an accident-by-accident basis, and if the frequency-consequence products summed over the locus of points representing the radiological risk acceptance guideline is less than the magnitude of the quantitative safety goal for latent cancer fatalities, then the sum of the calculated frequency-consequence products, or the total risk, will be less than the safety goal.

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It is necessary to recall that the DOE safety goal for prompt fatalities from an accident is intended to be applied to an average individual located near the site boundary of the facility. The safety goal for latent cancer fatalities (long term effects) is intended to be applied to a population within a specific distance from the facility site boundary of concern which is exposed to the radioactive material released from an accident. This safety goal also is expressed in terms of an average individual in the population.

However, the radiological consequences from the Radiological Risk Acceptance Guidelines are in terms of the consequences to a maximum site boundary individual. The maximum-exposed off-site individual is a hypothetical individual located at the site boundary in the direction of the worst-case atmospheric dispersion factor who generally remains at the plume centerline for the duration of a postulated release. Consequently, in making comparisons among the maximum exposed individual of the risk acceptance guidelines and the average individual of the DOE safety goal, it is necessary to establish a common basis so that the risks are comparable. For example, the consequences to the maximum site boundary individual need to be converted to the consequences to an average individual located in the population of concern. The average individual consequences then need to be expressed in terms of latent cancer fatalities per person. Specific unit conversions will be provided as necessary.

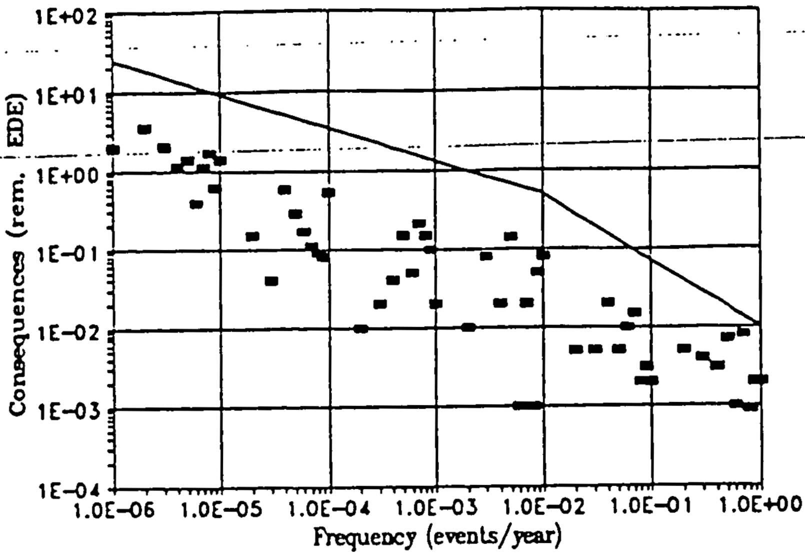

Figure 3. Example: Arbitrary Set of Risk Dominant Accident Sequences

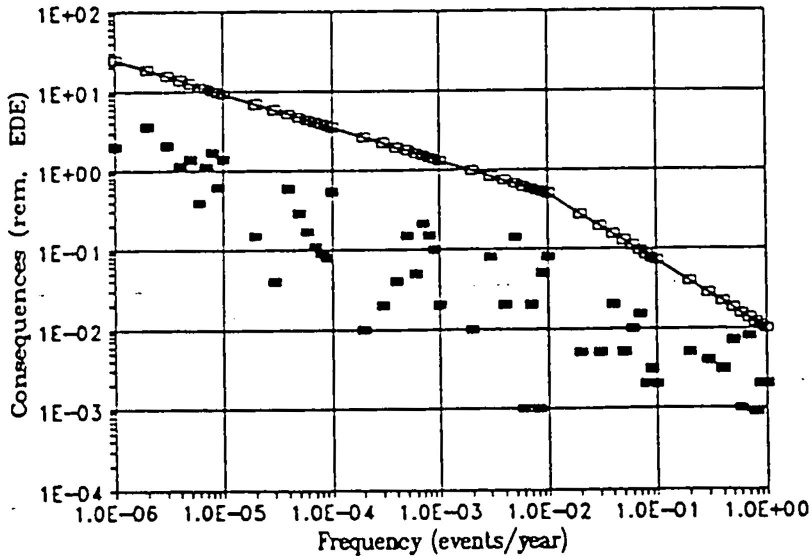

Figure 4. Example: Risk Dominant Accidents Compared To Guidelines

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C. DISCUSSION OF THE PROMPT FATALITY SAFETY GOAL

The proposed off-site Risk Acceptance Guidelines, shown in Figure 1, limit the effective dose equivalent (EDE) to the maximally exposed off-site individual to 25 rem at a frequency of 1×10^{-6}/yr, and well below 25 rem for events with higher frequency. The maximally exposed individual dose is based on worst-case meteorology (meteorology which would not be exceeded 95% of the time). It is generally accepted by the health physics community that acute whole body radiological doses up to 100 rem will not result in prompt fatalities. The LD-50 dose is generally taken to be between 200 to 400 rem.

Thus, limiting the maximally exposed individual to less than 25 rem at the site boundary ensures that there will be no prompt fatalities within one mile from the site boundary. Therefore, the DOE safety goal for prompt fatalities will always be met if the off-site risk acceptance guidelines are met.

D. DISCUSSION OF THE LATENT CANCER FATALITY SAFETY GOAL

There are two conversion factors for the consequences that need to be considered. First, the radiological dose to the maximum exposed individual at the site boundary needs to be converted to the radiological dose consequences to an average individual in the population within 10 miles of the facility, but beyond the site boundary. This conversion factor can be evaluated by either exact calculations or the dispersion coefficient, χ/Q. The conversion factor is expressed as the ratio of the radiological dose to (or χ/Q for) a maximum exposed individual at the site boundary to the radiological dose to (or χ/Q for) an average individual in the population within 10 miles of the facility.

In either case, the ratio ranges from $1 \times 10^{+3}$ to $1 \times 10^{+4}$ depending on the exact site meteorological conditions and the distance of the facility to the site boundary. Thus, the radiological dose to the average individual in the 10-mile annulus around the site will be at least a factor of 1×10^{-3} lower than the radiological dose to the maximum individual at the site boundary.

Second, the International Commission on Radiological Protection (ICRP) in ICRP Publication 60 (ICRP 1990) noted that the probability of fatal cancer after low dose, low dose-rate irradiation in a population of all ages is 5×10^{-2} fatalities/Sv (or 5×10^{-4} fatalities/rem). Thus, the radiological dose (rem) for an average individual in a population can be converted to latent cancer fatalities using the above conversion factor.

The task now is to demonstrate that the risk acceptance guidelines provided in Figure 1 satisfy the criteria necessary for the total risk to be less than the safety goal with the two conversion factors that have been developed. However, it is instructive to first consider the "peg points" given in the risk acceptance guidelines of 25 rem (EDE) at 1×10^{-6}/year, 0.5 rem (EDE) at 1×10^{-2} /year, and 0.01 rem at 1/year. These "peg points" do not represent the total risk acceptance curve, but the results will illustrate the step-wise calculations required. The results are given in Table 1.

Table 1. Conversion of Risk Acceptance Guideline "Peg Points"

Frequency (per year)	Maximum site boundary dose (rem, EDE)	Maximum to average ratio	Ten-mile annual average dose (rem /person)	Probability of fatal cancer (fatalities/rem)	Average individual risk (LCF/person-year)
1 E-06	25	1 E+3	2.5 E-02	5 E-04	1.25 E-11
1 E-02	0.5	1 E+3	5 E-04	5 E-04	2.5 E-09
1	0.01	1 E+3	1 E-05	5 E-04	5.0 E-09

As expected, the largest contribution to the total risk comes from the high frequency events. This fact raises an interesting point. In the analysis of an activity or facility, if there were to be a large number of events with frequency of occurrence in the range of once per 100 years (1×10^{-2}/year) to once per year (1/year), then the risk of the activity may not meet the DOE safety goal. The point is, the number of accident sequences considered must be realistic and finite for either a direct comparison of the total risk to the safety goal, or for a comparison of the risk acceptance guidelines to the safety goal.

Many combinations of frequency groups and accident sequences per frequency group were evaluated for the radiological risk acceptance guidelines. In all cases if the number of accident sequences was of the order of 1000, then the magnitude of the sum over F_j, C_{jM}, and L_j was roughly equal to or less than 2×10^{-6} latent cancer fatality/person-year. As noted earlier, the number of accident sequences considered must be realistic and finite for a comparison of the risk acceptance guidelines to the safety goal.

Thus, if the calculated frequency and consequence for each individual accident in an analysis of an activity or facility are compared with the risk acceptance guidelines, and if the radiological dose for each accident is below the guideline for the appropriate frequency, then the DOE safety goal for latent cancer fatalities will be satisfied.

It should be noted that requiring all accident sequences to be within the frequency-consequence space defined by the risk acceptance guidelines, and having a realistic number of accident sequences, is sufficient to show compliance with the DOE safety goal but it is not a necessary condition. There may be an infinite number of sets of frequency-consequence pairs that may not be within the risk acceptance guidelines, but may satisfy the DOE safety goal for latent cancer fatalities. The preceding discussion only demonstrates that a finite number of risk dominant accident sequences which satisfy the risk acceptance guidelines is a sufficient condition to show compliance with the DOE safety goal for latent cancer fatalities.

It should also be noted that the DOE safety goal for latent cancer fatalities applies to operations as well as to abnormal or accident conditions. The risk to the public due to routine emissions from normal operations is typically small in comparison with the risk due to postulated accidents. For example, the average per capita dose from all Hanford Site operations in 1991 within 50 miles of the site boundary was 0.002 mrem (PNL 1991). This would correspond to 1×10^{-9} latent cancer fatalities per average individual or 0.05 % of the DOE safety goal based on the ICRP 60 dose conversion factor of 5×10^{-4} LCF/rem (ICRP 1990). The value of 0.002 mrem is due to the total routine emissions from all facilities within the Hanford Site. Therefore, the latent cancer fatality risk from any one facility on the Hanford Site would be lower than 0.05 % of the DOE safety goal.

VII. APPLICATION

A. GENERAL

The radiological risk acceptance guidelines are to be used in the process of implementing the requirements of DOE Order 5480.23 "Nuclear Safety Analysis Reports". The guidelines are one element in the continual evaluation of the results of postulated accident sequences to determine that the risks are within the bounds set by the DOE Nuclear Safety Policy. Because safety analysis is an iterative process which continues throughout the design and life of an activity, the continual comparison of individual accident risks to the risk acceptance guidelines permits identification and incorporation of alternatives in the design or operation to maintain an acceptably low risk. In the early design phase, information available for definitive risk estimates may be minimal. As the design and analysis progress, more detailed information is developed which permits more refined risk estimates. The advantage of an evolving accident-by-accident analysis and comparison with an appropriate criteria is that one is assured that when the final analysis is completed, where the total risk can be quantified, the total risk will be acceptable and in compliance with the DOE Nuclear Safety Policy.

The risk acceptance guidelines are suitable for evaluating the risk presented by existing and new activities, as well as modifications to existing activities. The techniques available to limit risk in each case are different. For new or modified facilities or processes, changes can be made to the design or process to modify the resulting risk. The analyst can rapidly evaluate the effect of the changes. For risks which cannot be reduced in this manner, the alternatives of imposing administrative controls on the process are available. For existing facilities, which may not be suitable to structural or process design changes, the analyst has the alternative of imposing administrative controls to limit the total risk of the activity.

These risk acceptance guidelines only apply to the analysis of risks from radiological exposures. Other risks which are associated with exposure to hazardous materials must be analyzed separately. Risk based guidelines are being prepared for evaluating accidents related to hazardous material exposure.

The Off-Site Risk Acceptance Guidelines were developed for and are applicable to analyses which estimate the risks to the off-site public. The On-Site Risk Acceptance Guidelines were developed for and are applicable to analyses which estimate the risks to collocated workers and other on-site individuals who have been provided training as specified in DOE Order N5480.6 (Radiological Control Manual), Chapter 6. Members of the public who may be on-site at visitor centers or on public thoroughfares are addressed in Section F.

B. GENERAL ANALYSIS PROCESS

1. DEVELOPMENT OF ACCIDENT SEQUENCES

In considering accident sequences, the goal is to identify and develop a finite set of risk dominant accidents. That is, for the set of representative accidents that may have been identified, the sum of the individual accident risks (product of frequency-consequence pairs) should approach the total risk due to all accidents. In addition, sets of accidents whose individual risks may not be significant but whose sum of individual risks is significant should be included. In accordance with DOE Order 5480.23, both design-basis accidents (DBAs) and beyond-design-basis accidents should be included.

A design-basis accident is an event for which a design feature has been provided so that the results of the accident will not have unacceptable adverse effects on the public. In analyzing a design-basis accident, the design feature is assumed to function during the accident. For example, a design basis earthquake (DBE) is the earthquake of maximum severity which the facility will withstand. The facility is not assumed to withstand an earthquake beyond the design basis.

Developing a representative set of accident sequences usually consists of; (a) identifying as many potential accidents as possible, (b) selecting representative and risk dominant accidents from this set, and (c) describing the sequence of steps that represent the accident progression for each representative accident. Each of these basic steps will be discussed.

Hazards Identification

The first step in developing postulated accident scenarios is to identify as many accidents as possible without regard to their importance (occurrence, consequence, or frequency). Many formalized procedures exist for identifying hazards, especially those used successfully in the chemical process industries (e.g., CCPS 1985, 1992). Examples, in order of increasing complexity, are checklists, hazard indices, "what if" considerations, Preliminary Hazard Analysis (PHA), Failure Modes and Effects Analysis (FMEA) coupled with human reliability analysis (HRA), and hazards and operability studies (HAZOPS). Some procedures are more appropriate in certain circumstances than are others. For example, a PHA appears especially suitable early in the design process. Thus, the choice of which hazards identification procedure to use depends on the situation.

Choosing Representative Accidents

The next step is to reduce the list of accidents identified in the hazards identification to a much smaller list of accidents that require analysis. The number of accidents that should be analyzed depends on the level of detail which the facility merits. CCPS (1989, pp. 24-25) lists one approach appropriate to accident reduction. The approach involves first eliminating those accidents too minor to be of concern. Redundant or very similar events, or events with very similar consequences, are then combined into groups. The process then combines approximately similar events into subsets, followed in turn by replacing subsets with equivalent events. Finally, depending on the level of detail required, representative events or bounding events might be selected for analysis. For a facility which might merit a detailed analysis, the entire list of equivalent events might be selected. The selection of events should be tailored to the facility's complexity and its ability to impair public health. The intent is to identify a set of representative accidents which are the risk dominant accidents. The set of representative, risk dominant accidents should include those with initiating events resulting from natural phenomena, process upsets, and human error.

Accident Progression

The third step is to describe the progression of each representative accident from the initiating event through to the release of material to the environment and the consequences to the receptor. Formal procedures have been developed for following accident progression. Some procedures suitable for hazard identification are also useful here, such as HAZOPS and FMEA-HRA. More sophisticated methods such as fault tree analysis, event tree analysis, or combinations of event tree and fault tree analysis, may also be appropriate. Different approaches may be more suitable for various situations. Computer software is available to aid in the development of accident scenarios.

2. DETERMINATION OF ACCIDENT FREQUENCIES AND CONSEQUENCES

To this point, development of the representative accident scenarios has been entirely qualitative. The next step is to quantify the accident frequencies and consequences. That is "How likely are the representative accidents?" and "What are their consequences?

The quantification of accident sequences also usually progresses in steps, with the amount of detail required depending again on the facility's complexity and its ability to impair public health. The first step is purely deterministic and only involves a conservative evaluation of the consequences. The worst-case accident scenario is considered with conservative assumptions. For example, the worst-case accident scenario may consider the entire radioactive material process inventory, reduced by a realistic fraction to consider material form, released as the source term. If the radiological dose consequences of this postulated worst-case accident is smaller than 10 mrem, (the lowest consequence for the highest frequency considered in the risk acceptance guidelines) then it may be assumed that the consequences of all other credible accidents are lower than the risk acceptance guidelines for all frequencies. No further analysis would be necessary.

The next step would be to consider some of the most representative accidents and to evaluate their consequences using conservative assumptions. The likelihood of the accident should be estimated. Because of the uncertainty in the estimated accident frequency, it is appropriate at this stage to use qualitative frequency groups, such as likely (1/year to 1 x 10^{-2}/year), unlikely (1 x 10^{-2}/year to 1 x 10^{-4}/year), extremely unlikely (1 x 10^{-4}/year to 1 x 10^{-6}/year), and incredible (less than 1 x 10^{-6}/year). If the consequences of these accidents do not exceed the risk acceptance guidelines at the high-frequency end of the appropriate frequency range for each accident, then no further analysis would be necessary.

If in either case the consequences exceeded the risk acceptance guidelines then further analyses and/or design changes would be required. Again the amount of detail required depends on the facility's complexity and its ability to impair public health.

Frequency Determination

Often a more accurate quantification of the accident frequencies is required. For some situations, accident data bases or equipment reliability data bases can be used to provide accident frequencies for simple situations. However, there are more sophisticated methods which can be used to quantify accident frequencies. These methods are used in a variety of procedures such as Probabilistic Risk Assessment (PRA), Probabilistic Safety Assessment (PSA), and Quantitative Risk Assessment (QRA). Sufficient literature exists which describes these methods in detail such that only a quick summary is required here.

The methods use event trees, fault trees, and/or combinations of event trees and fault trees to thoroughly define all potential accident scenarios. By assigning probabilities to each branch point in the event tree or fault tree, the trees can be solved (usually with available computer codes) to determine the frequency for each accident scenario. These methods also can consider the uncertainty associated with each branch point probability or initiating frequency to determine the uncertainty of the accident frequency.

Consequence Determination

The accident results in the actual release of radioactive material, toxic material, or energy which may cause injury or loss. The consequences considered here are only those resulting from the release of radioactive material. The progression of the accident analysis usually follows the following steps. First, the radioactive material inventory is identified, (2) the radioactive material at risk due to the process is determined, (3) the energy source available or generated as a result of an accident which can release material is identified or calculated, (4) the amount of radioactive material released to the environs as a result of the accident is determined, (5) the transport of radioactive material from the release point to a receptor is modeled, and (6) the radiological consequence of the material intercepted by a receptor is determined. An event tree, fault tree or event tree - fault tree combination is usually created which defines each accident sequence. The amount of radioactive material following each accident sequence is then determined. When the radioactive material encounters vulnerable individuals, a consequence follows. The sum of the individual consequences is the total consequence of the postulated accident.

C. COMPARISON OF ACCIDENT CONSEQUENCES AND FREQUENCIES WITH RISK ACCEPTANCE GUIDELINES

In comparing the calculated accident consequences and frequencies with the risk acceptance guidelines, it is important to consider the uncertainty associated with both the consequence and frequency calculations. For example, in the early stages of a project design or when sufficient information is not available to completely quantify the accident frequencies, qualitative frequency groups may be used. Appropriate qualitative frequency groups are: likely (1/year to 1×10^{-2}/year), unlikely (1×10^{-2}/year to 1×10^{-4}/year), extremely unlikely (1×10^{-4}/year to 1×10^{-6}/year), and incredible (less than 1×10^{-6}/year). Also, conservative assumptions may be required in order to obtain an estimate of the accident consequences.

As the design progresses and/or more information becomes available, the frequencies and consequences can be determined with greater certainty. However, the uncertainty in both the calculated consequences and frequencies may still be substantial and needs to be considered as discussed below. Several situations in the comparison of the calculated accident consequence-frequency pair with the risk acceptance guidelines need to be considered.

Calculated Consequence-Frequency Pair Lies Below Guidelines

If the calculated consequence-frequency pair, as graphed in a two dimensional consequence-frequency space, lies well below the risk acceptance guidelines then no more consideration may be required for the specific accident scenario. It is advisable to identify the assumptions and restraints of the analysis, specific design features, or administrative controls considered in the analysis which are responsible for the low calculated accident consequence and/or frequency. The design features, assumptions, restraints or controls may be the basis for safety class systems, and/or technical safety requirements. Even if the calculated consequence-frequency pair lies significantly below the risk acceptance guidelines, a qualitative examination, in the philosophy of ALARA, should be made to confirm that all reasonable actions have been taken to reduce potential radiation exposure.

Calculated Consequence-Frequency Pair Lies Above Guidelines

If the calculated consequence-frequency pair, as graphed in a two dimensional consequence-frequency space, lies above the risk acceptance guidelines then more consideration of the specific accident scenario is required. The nature of the consideration may be one of the following, or a combination of the following, steps.

First, the analyst may be able to refine the calculations, reassessing assumptions used and their conservatism. By this process either the calculated consequence and/or calculated frequency of the accident scenario may be reduced until it is demonstrated that the calculated accident consequence-frequency pair lies below the risk acceptance guideline.

Second, it may be necessary to include a preventative or mitigative feature in the facility or process design to either prevent the postulated accident from occurring (reduce the likelihood of occurrence), or mitigate the consequences of the postulated accident (reduce the consequences) should the accident occur. With the preventative or mitigative design feature added to the facility or process, a re-analysis of the postulated accident is completed. This iterative process is continued until it is demonstrated that the calculated accident consequence-frequency pair lies below the risk acceptance guideline.

Third, there may be some design changes (other than adding preventative or mitigative features) which may be made to the facility or process which will either reduce the frequency and/or consequence of the postulated accident. Again, a re-analysis of the postulated accident is completed with the added design features included until it is demonstrated that the calculated accident consequence-frequency pair lies below the risk acceptance guideline.

Fourth, it may be possible to reduce either the postulated accident frequency and/or consequences by imposing administrative controls or procedures to the facility or process. The administrative controls or procedures are chosen such that the frequency and/or consequence of the postulated accident are reduced. A re-analysis of the postulated accident is completed taking into account the administrative controls or procedures until it is demonstrated that the calculated accident consequence-frequency pair lies below the risk acceptance guideline.

Regardless of which step, or combination of steps were used to eventually demonstrate compliance with the risk acceptance guidelines, it is advisable to identify the design feature or features, administrative controls or procedures, or analytical assumptions or restraints which were necessary to obtain compliance. These design features should be identified as systems, components or structures important to safety. The administrative controls or procedures may be part of a Technical Safety Requirement.

Calculated Consequence-Frequency Pair Lies Near Guidelines

If the calculated consequence-frequency pair, as graphed in a two dimensional consequence-frequency space, lies below, but near the risk acceptance guidelines then special consideration is required. The same four steps, or combination of steps, discussed above could be used to further reduce either the calculated consequence or the calculated frequency.

If uncertainties in both the frequency and consequence of an accident scenario are known, then confidence limits can be used to determine if the calculated consequence-frequency pair of an accident are far enough below the risk-acceptance guidelines. For example, one criterion would be that the accident should meet the risk-acceptance guidelines at the 95% confidence limit (two standard deviations above the mean, if the frequency or consequence is normally distributed). This means that the 95% confidence limit in both the consequence and frequency should be the point which lies below the risk-acceptance guidelines.

If, however, either the frequency or the consequence is not known with sufficient certainty that confidence limits can be determined, then the following guide may be applied. The best estimate of the accident consequence should be at least one decade below the risk-acceptance guidelines for the best estimate of the frequency.

Recall that requiring the consequence-frequency pair for all accident sequences to be below the risk acceptance guidelines is a sufficient condition to show compliance with the DOE safety goal, and hence, acceptable risk. However, the As Low As Reasonably Achievable (ALARA) philosophy should also be implemented. Effort should be made to reduce the calculated consequence and frequency for each postulated accident to as low as reasonably achievable.

In general, the closer the calculated consequence-frequency pair approaches the limiting risk acceptance guidelines, the more quantitative the analysis should be, including uncertainties. The approach is to assure that the individual accident risk is truly bounded by the applicable guidelines.

D. CALCULATIONAL CONSIDERATIONS

There are many steps involved in evaluating the consequences of accident sequences. The techniques and methods are frequently left to the analyst, but some steps have become so specific that precise techniques are required to be used. Calculational considerations are required in identifying the radioactive material inventory, inventory form, and dispersibility; the accident event propagation time; the action of mitigation processes; the material release time, etc.

Accident Release Time

Accident release times should be realistic and applicable to the accident being considered. For example, for the on-site or collocated workers the release time should be consistent with the time involved in the accident progression, the time required to identify that an accident has occurred, and time required to move and/or evacuate personnel. For the off-site public, the release time should be consistent with the time required for the radioactive plume to reach the receptor location and the time duration of the accident.

Meteorological Data

Site specific meteorological data should be used if available. Ninety-five percent meteorological conditions should be used for conservatism. This means that only 5% of the time will the meteorological conditions be worse than those used. If site specific meteorological data is not available, NRC Reg Guide 1.145 conditions of F class stability and a wind speed of 1 meter/sec should be used.

Population Distribution Data

The radiological risk acceptance guidelines used the maximum exposed individual concept to estimate radiological dose consequences. One can use population distributions in the one-mile or ten mile sectors. Current census data and any identifiable significant changes should be included in the population distribution. Population projections should be considered.

E. LIMITATIONS OF GUIDELINES

The radiological risk acceptance guidelines are to be used in an accident-by- accident comparison. The guidelines are not related to the total risk of a facility. It has been assumed that the risk

dominant accident sequences have been identified and used in the analysis. Therefore, proper consideration should be given to be sure that the accident set chosen for analysis is as complete as possible, and that at least the risk dominant accidents are analyzed.

Compliance with the radiological risk acceptance guidelines provides a sufficient condition to demonstrate compliance with the DOE Nuclear Safety Policy, but not a necessary condition. For some facilities, there may be sets of consequence-frequency pairs that may not be within the risk acceptance guidelines, but the total risk would still satisfy the DOE safety goals. To demonstrate compliance with the DOE safety goals for this situation, a complete risk assessment may be required.

Special consideration should be given to the uncertainties in both the calculated frequencies and calculated consequences. The uncertainties are especially important in the early design phase when sufficient information is not available to quantify the frequencies or be very precise in the consequences. The use of consequence-frequency ranges is acceptable.The radiological risk acceptance guidelines are only one of several guidelines, criteria, standards and/or policies that are used to make decisions important to the safety of the public and DOE workers. Care should be taken not to infer more information from the guidelines than is really available based on the facility process or design.

F. APPLICATION TO ON-SITE PUBLIC AND FACILITY WORKERS

The DOE nuclear safety policy only considers the risk to the off-site public. There are instances in which members of the public are on DOE sites. That is, members of the public may be travelling on a highway or river which runs through a DOE site or be at a visitors center or other point of interest. ALARA requires that these individuals should receive the minimum exposure practical. The premise is that the on-site public should be afforded equivalent protection as the off-site public even though they are much closer to a potential release point.

The factors which control the risk to the on-site public are the distance from the source, location of the individuals relative to the direct plume, time available for institution of emergency actions, and length of time the individuals are in the direct plume. The possibilities available to limit radiation exposure of the on-site public include access restrictions and evacuation or relocation instituted by emergency plans. Engineered safety features are installed to protect the off-site public. Emergency plans, which include the provision for evacuation or relocation, are administrative safety features available for use to protect the on-site public. The assumptions used regarding the effectiveness of emergency plans (i.e. the time required to initiate or complete an action) as administrative controls should be supported by adequate data to support them. If the set of safety features and controls are not sufficient to provide the required protection, then other restrictions on public access or operational limits may be required.

Specific guidelines are not provided for facility workers. Facility worker risks are controlled by training and awareness of radiological risks, use of approved procedures, emergency response training, and the application of ALARA in operations.

G. ALTERNATIVE METHODS

It is always possible that for any facility and/or process a complete risk assessment could be completed. The calculated total risk could then be compared directly to the two quantitative safety goals of the DOE Nuclear Safety Policy. Thus, protection of the health and safety of the public could be directly demonstrated. The risk assessment would also have to consider the risk to collocated workers.

The risk acceptance guidelines used as consequences the radiological consequences to the maximum exposed off-site or maximum exposed on-site individual, in rem, effective dose equivalent. Other units for the radiological consequences could also be used. Other receptor definitions could also be used, such as an average individual, annual average individual, or the average individual in a sector. The important point is that in comparing radiological dose consequences to the Nuclear Safety Policy that appropriate conversion factors be identified and used. The use of the concept of a maximum exposed off-site or maximum exposed on-site individual provides a degree of conservatism that seems appropriate when making a comparison with guidelines.

VIII. OTHER CONSIDERATIONS

A. MULTIPLE FACILITIES

The risk acceptance guidelines were developed to apply to individual release accidents in a specific facility SAR. The guidelines do not address releases from multiple facilities. The issue is best illustrated by the potential for accidents that could affect several facilities, resulting in off-site dose consequences from each facility that would be additive. M&O contractors and DOE should ensure that accidents at closely-located facilities do not result in unacceptable radiological off-site dose consequences. M&O contractors and DOE should also ensure that closely-sited facilities and individual facilities do not have a large number of potential accidents in the anticipated frequency range (1/yr to E-2/yr) that could result in unacceptable consequences.

B. CHANGING REGULATORY REQUIREMENTS

The risk acceptance guidelines are based on current guidelines and regulations that were carefully reviewed. Although future regulations may be more restrictive (e.g. Team B Modernization Report), they are not anticipated by this document. The guidelines use conservative values and assumptions.

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APPENDIX

Table 1. Whole Body Radiation Standards for Routine Exposure of the Public

DOSE	APPLICABILITY	REFERENCE
100 mrem/yr (higher dose allowed if average over 5 years is less than 100 mrem/yr)	"Limit for public exposure"	ICRP 60 (1991)
100 mrem/yr 500 mrem/yr	Continuous or frequent Infrequent exposure	NCRP 91 (1987)
0.1 rem/yr 25 mrem/year	Individual Airborne effluents	40 CFR 61, Subpart II (EPA) (1990)
25 mrem/yr	Uranium fuel cycle	40 CFR 190 (EPA) (1987)
4 mrem/yr	Releases to community drinking water systems	40 CFR 141 (EPA) Subpart A (1987)
25 mrem/yr	Discharges and direct radiation from waste disposal for 1000 years	40 CFR 191 (EPA) Subpart A (1987)
25 mrem/yr	All potential pathways from waste disposal for 1000 years	40 CFR 191 (EPA) Subpart B (1987)
25 mrem/yr	Uranium mill tailings	40 CFR 192 (EPA) 1987
25 mrem/yr	Off-site exposure from low level waste storage and disposal	40 CFR 193 (EPA) DRAFT (1987)
4 mrem/yr	Groundwater contaminated by disposal activities	40 CFR 193 (EPA) DRAFT (1987)
100 mrem/yr	"Reference level" for each licensee	10 CFR 20 (1991)
25 mrem/yr	"Uranium mill tailings	10 CFR 40 (NRC) (1988)
3 mrem/yr	Design Objectives for light water reactor liquid effluents	10 CFR 50 Appendix I (NRC) (1986; still effective?)
5 mrem/yr	Design objectives for light water reactor gaseous effluents	10 CFR 50 Appendix I (NRC) (1986; still effective?)

DOSE	APPLICABILITY	REFERENCE
625 mrem/yr	License requirement (releases from near-surface land disposal)	10 CFR 61 (NRC) (1988)
500 mrem/yr	Inadvertent intruders on near-surface disposal site	10 CFR 61 (NRC) (1988)
25 mrem/yr	Public dose near low level waste storage or disposal facilities	DOE Order 5820.2A (1988)
100 mrem/yr	Continuous exposure to inadvertent intruder	DOE Order 5820.2A (1988)
500 mrem/yr	Single acute exposure to inadvertent intruder	DOE Order 5820.2A (1988)
100 mrem/yr (500 mrem/yr if average lifetime dose less than 100 mrem/yr)	All exposure modes, all DOE sources of radiation; "potential public dose"	DOE Order 5400.5 (1990)
10 MREM/YR	Airborne emissions only, all DOE source applies "off-site where members of the public reside or abide"	DOE Order 5400.5 (1990)
25 mrem/yr	All exposure modes from storage of spent nuclear fuel, high-level and TRU wastes at disposal facilities	DOE Order 5400.5 (1990)
4 mrem/yr	Drinking water pathway, all DOE sources	DOE Order 5400.5 (1990)

Notes

1. NCRP Recommendations on control of sources:

In recognition of the possibility that all members of the public could receive significant exposure from a number of different sources, the NCRP has developed the following recommendation - if the potential exists for an individual to exceed 25 % of the annual effective dose equivalent limit from exposure attributable to a single site, then the site operator should ensure that the annual effective dose to maximally exposed individuals from all sources would not exceed 100 mrem on a continuous basis. This recommendation assumes that significant exposure from more than four sources is unlikely and essentially would provide a limit on annual effective dose equivalent of 25 mrem per source.

2. "A substantial number of environmental radiation standards may limit risks at levels generally regarded as negligible by the public." (D.C. Kocher, "Review of Radiation Protection and Environmental Radiation Standards for the Public," Nuclear Safety, Vol. 29, No. 4, Oct-Dec 1988.)

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Table 2. Whole Body Radiation Standards for Routine Exposure of the Workers

DOSE	APPLICABILITY	REFERENCE
5 rem/yr	All pathways	ICRP 26 (1977)
5 rem\yr	External	ICRP 26 (1977)
5 rem in one event and/or 25 rem in a lifetime	Planned special exposure	ICRP 26 (1977)
0.5 rem in 2 months to fetus	Women of reproductive capacity	ICRP 26 (1977)
5 rem CEDE	Worker	Radiation Protection Guidance to Federal Agencies for Occupational Exposures
5 rem/yr CEDE	Worker	10CFR 20 (1991)
0.5 rem during gestation	Pregnant women/fetus	10CFR 20 (1991)
5 rem/yr gamma	Worker	30 CFR 57 (Office of Mine Safety & Health)
5 rem/yr	Worker	DOE ORDER 5480.11 (1990)
0.5 rem	Unborn child	DOE Order 5480.11 (1990)
10 rem	Planned special exposure	DOE Order 5480.11 (1990)
100 rem/yr	Worker under 18 and students	DOE Order 5480.11 (1990)
100 mrem/yr	Public entering controlled area	DOE Order 5480.11 (1990)
100 rad	Saving a human life	DOE Order 5480.11 (1990)
10 rem/yr	Recovery of deceased victim	DOE Order 5480.11 (1990)
10-25 rem/yr	Protection of health and property	DOE Order 5480.11 (1990)
2 rem/yr, averaged over defined periods of 5 years (less than 5 rem in any single year)	Worker	ICRP 60 (199)
200 mrem at surface of abdomen during gestation	Pregnant worker	ICRP 60 (199)
50 rem	Emergency	ICRP 60 (199)
more than 50 rem	Life-saving	ICRP 60 (199)
0.5 rem/9 months	Pregnant female	NCRP 91
5 rem/yr maximum	Worker	NCRP 91
70 rem/yr lifetime cap	Worker	NCRP 91
Lifesaving 100 rem	Worker aware of risk	NCRP 91
10 rem single exposure	Worker	NCRP 91

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